KHNPDCDRAIsPEm Resource

From:	Ciocco, Jeff
Sent:	Tuesday, October 27, 2015 11:54 AM
To:	apr1400rai@khnp.co.kr; KHNPDCDRAIsPEm Resource; Harry (Hyun Seung) Chang; Andy Jiyong Oh; Christopher Tyree
Cc:	Schmidt, Jeffrey; McKirgan, John; Steckel, James; Lee, Samuel
Subject:	APR1400 Design Certification Application RAI 273-8365 (15.02.08 - Feedwater System Pipe Breaks Inside and Outside Containment (PWR))
Attachments:	APR1400 DC RAI 273 SRSB 8365.pdf

KHNP,

The attachment contains the subject request for additional information (RAI). This RAI was sent to you in draft form. Your licensing review schedule assumes technically correct and complete responses within 30 days of receipt of RAIs. However, KHNP requests, and we grant, the following response time for the RAI questions. We may adjust the schedule accordingly.

15.02.08-1: 45 days
15.02.08-2: 30 days
15.02.08-3: 30 days
15.02.08-4: 30 days
15.02.08-5: 45 days
15.02.08-6: 45 days

(Please note, for administrative reasons only, draft RAI 8334 became RAI 8365.)

Please submit your RAI response to the NRC Document Control Desk.

Thank you,

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REQUEST FOR ADDITIONAL INFORMATION 273-8365

Issue Date: 10/27/2015
Application Title: APR1400 Design Certification Review – 52-046
Operating Company: Korea Hydro & Nuclear Power Co. Ltd.
Docket No. 52-046
Review Section: 15.02.08 - Feedwater System Pipe Breaks Inside and Outside Containment (PWR)
Application Section:

QUESTIONS

15.02.08-1

10 CFR Part 50 Appendix A, GDC 31 requires that the reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. SRP 15.2.8 provides acceptance criteria that the RCS and main steam system pressures stay below 110 percent of the design values for low probability events and 120 percent for very low probability events.

In DCD Section 15.2.8.3.1, Evaluation Model, assumptions regarding the affected steam generator heat transfer characteristics are made based on liquid mass inventory. The mass difference where the heat transfer area decreases from the design value to zero is defined by the change in liquid mass. The change in affected steam generator liquid mass is assumed to be 0 kg as a modeling conservatism. A change of 0 kg implies that the affected steam generator heat transfer area is also zero and hence no heat transfer would occur at or very near time zero. According to DCD Table 15.2.8-2, the affected steam generator loses heat transfer 27.50 seconds following the break. Delaying the loss of the affected steam generator heat transfer could decrease peak RCS pressure. Explain how the modeling assumption of zero heat transfer area for a 0 kg change in affected steam generator mass is consistent with losing the affected steam generator heat transfer capability at 27.50 seconds.

15.02.08-2

10 CFR Part 50 Appendix A, GDC 31 requires that the reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. SRP 15.2.8 provides acceptance criteria that the RCS and main steam system pressures stay below 110 percent of the design values for low probability events and 120 percent for very low probability events.

DCD Tier 2, Table 15.2.8-1, ”Initial Conditions for Limiting Case Feedwater Line Break,” provides the initial conditions used for the steam generator level (97,046 kg). It is unclear from the DCD the basis for the initial steam generator level value chosen. The initial steam generator level affects the heat removal capability of the unaffected generator as well as the timing at which the affected steam generator loses heat removal capability. Explain the basis for the initial level, why it is conservative, and revise the DCD as necessary.

15.02.08-3

10 CFR Part 50 Appendix A, GDC 31 requires that the reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. SRP 15.2.8 provides acceptance criteria that the RCS and main steam system pressures stay below 110 percent of the design values for low probability events and 120 percent for very low probability events.

In DCD Tier 2, Section 15.2.8.3.1, "Evaluation Model," the applicant states that no single failures would affect the primary to secondary heat transfer. Primary to secondary heat transfer are not only potentially affected by single failures but could also be affected if hot leg saturation exists at any time during the transient. Evaluate the potential for hot leg saturation and revise the DCD to include a discussion regarding hot leg saturation.

15.02.08-4

The radiological consequences calculated to show compliance with 10 CFR 50.34 is dependent on the number of potentially failed fuel rods. In Section 15.2.8.5.2, "Input Parameters and Initial Conditions," a statement is made that no fuel damage is postulated for the feedline break analysis. This statement is based on the DNBR remaining above 1.29 minimum DNBR limit. According to DCD Section 15.2.8.3.2, "Input Parameters and Initial Conditions," the fuel integrity analysis uses an initial pressurizer pressure of 2,325 psia. From DCD Table 15.0-3, "Initial Conditions," the low pressurizer pressure value is 2,175 psia. What is the basis for using a higher pressurizer pressure than the lower end of the initial conditions as the DNBR decreases with lower pressures?

15.02.08-5

The radiological consequences calculated to show compliance with 10 CFR 50.34 is dependent on the number of potentially failed fuel rods. Examining DCD Tier 2, Figure 15.2.8-1, "Main Feedwater Line Break with Concurrent LOOP: Maximum RCS Pressure vs. Break Area," the staff noted that no peak pressure was plotted for a 0.0093 m^2 break and hence was not able to determine based on the plot trend if the lowest peak RCS pressure case was used to evaluate DNBR. Revise Figure 15.2.8-1 to include the 0.0093 m^2 break data point. If the 0.0093 m^2 break peak pressure is higher than the 0.02 m^2 break size pressure explain why using the 0.0093 m^2 break peak pressure yields a conservative DNBR.

15.02.08-6

10 CFR Part 50 Appendix A, GDC 31 requires that the reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. SRP 15.2.8 provides acceptance criteria that the RCS and main steam system pressures stay below 110 percent of the design values for low probability events and 120 percent for very low probability events.

In Section 15.2.8.4.3, “Results” it is stated the maximum steam generator pressure increases for the offsite power available case relative to the LOOP case. Based on the DCD write-up it is unclear if a spectrum of break sizes were examined to yield the highest steam generator pressure with offsite power available. Provide a justification as to why the 0.0372 m^2 break size used in the LOOP case maximizes steam generator pressure with offsite power or revise the DCD to include the break size which maximizes steam generator pressure with offsite power available.

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